UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

May 10, 2007

Mr. Fred R. Dacimo
Site Vice President
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, NY 10511-0249

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT 3 - NRC INTEGRATED INSPECTION REPORT 05000286/2007002

Dear Mr. Dacimo:

On March 31, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Indian Point Nuclear Generating Unit 3. The enclosed integrated inspection report documents the inspection results, which were discussed on April 4, 2007, with Mr. James Comiotes and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations, and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, five findings of very low safety significance (Green) were identified. Four of these findings were also determined to be violations of NRC requirements. However, because of their very low safety significance, and because the findings were entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any of the NCVs in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Senior Resident Inspector at Indian Point Nuclear Generating Unit 3.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,
/RA/
Eugene W. Cobey, Chief
Projects Branch 2
Division of Reactor Projects

Docket No. 50-286
License No. DPR-64

Enclosure: Inspection Report No. 05000286/2007002
w/Attachment: Supplemental Information

cc w/encl:
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-286
License No.:	DPR-64
Report No.:	05000286/2007002
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	Indian Point Nuclear Generating Unit 3
Location:	450 Broadway, GSB Buchanan, NY 10511-0249
Dates:	January 1, 2007, through March 31, 2007
Inspectors:	D. Jackson, Senior Resident Inspector, IP3 B. Wittick, Resident Inspector, IP3 M. Cox, Senior Resident Inspector, IP2 G. Bowman, Resident Inspector, IP2 S. Barr, Senior Emergency Preparedness Specialist F. Arner, Senior Reactor Engineer H. Gray, Senior Reactor Engineer D. Orr, Senior Project Engineer J. Noggle, Senior Health Physicist A. Patel, Reactor Engineer J. Sullivan, Operations Engineer R. Cureton, Emergency Preparedness Specialist C. Hott, Project Engineer T. Nicholson, Senior Technical Advisor on Radionuclide Transport
Other:	J. Williams, Groundwater Specialist, U.S. Geological Survey
Approved by:	Eugene W. Cobey, Chief Projects Branch 2 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000286/2007-002; 01/01/2007 - 03/31/2007, Indian Point Nuclear Generating Unit 3; Maintenance Effectiveness, Permanent Plant Modifications, Refueling and Outage Activities, and Temporary Plant Modifications.

The report covered a three-month period of inspection by resident and region-based inspectors. Five Green findings were identified, four of which were also non-cited violations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process (SDP) does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC Identified and Self-Revealing Findings

Cornerstone: Initiating Events

Green. The inspectors identified a Green non-cited violation (NCV) of 10 CFR 50.65(b), in that, Entergy did not include the Indian Point Unit 3 trash rack structures within the scope of the maintenance rule monitoring program. Additionally, Entergy did not demonstrate the performance or condition of the trash racks was being effectively controlled through the performance of appropriate preventive maintenance such that the structure remained capable of performing its intended function. Entergy performed a cleaning of the trash racks to immediately address the lowered service water intake bay level, and they timed service water bay level monitoring to coincide with river low tide cycles. Entergy also entered this issue into the corrective action program as CR-IP3-2007-00453, and developed corrective actions to: modify the requirements for inspection and cleaning of trash racks based on component history and condition monitoring; modify guidance for service water bay level monitoring to be more effective; evaluate maintenance rule system scoping; develop procedural guidance for managing low service water bay levels; and implement a method for monitoring debris fouling of the trash racks.

The inspectors determined that this finding affected the Initiating Events cornerstone and was more than minor because it was similar to Example 7.d in Inspection Manual Chapter (IMC) 0612, Appendix E, "Examples of Minor Issues." Specifically, equipment performance problems were such that Entergy was unable to demonstrate effective control of the performance or condition of the trash racks through appropriate preventive maintenance as specified by 10CFR50.65(a)(2). The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that the finding was of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. (Section 1R12)

Green. A Green, self-revealing, non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified, in that, Entergy's work package failed to ensure that piping interference was correctly planned for and removed during modifications to the vapor containment and recirculation sumps. On March 9, 2007, during the sump modifications, a section of pipe was cut for interference removal which was different from the piping specified in the work package. This resulted in approximately 385 - 500 gallons of reactor coolant being discharged from the reactor loops into the recirculation sump where personnel were working. The cause of the improper pipe being cut was misidentification of the piping by work planners, followed by a failure of workers to follow steps in the work package that should have identified the work package inadequacy. Immediate corrective actions included a revision to the work package that subsequently welded a cap on the open piping leading from the reactor coolant drain tank to the work site, and plant configuration tags were placed on the residual heat removal interface valves (SI-864E and 864F) to isolate the work area. Entergy entered this issue into the corrective action program as CR-IP3-2007-01059, performed a root cause analysis, and conducted a human performance error review.

The inspectors determined that this finding was more than minor because it was associated with the Procedure Quality attribute of the Initiating Events cornerstone; and, it affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, cutting the wrong pipe resulted in the inadvertent draining of reactor coolant system inventory and increased the likelihood of a loss of inventory control. This finding was evaluated using Phase 1 of IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." The inspectors evaluated the plant conditions (cold shutdown, reactor coolant system open, refueling cavity less than 23 feet) in accordance with Checklist 3 of Appendix G, Attachment 1, and determined that the finding was of very low safety significance because it did not satisfy the criteria of Table 1 for a "Loss of Control," and the Checklist 3 criteria for maintaining adequate mitigation capability (Core Heat Removal Guidelines, Inventory Control Guidelines, Power Availability Guidelines, Containment Control Guidelines, and Reactivity Guidelines) were met.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because the work package used for interference removal was not accurate and did not ensure the correct section of piping was identified and appropriately controlled. (Section 1R17)

Green. A Green, self-revealing, non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified, in that, Entergy failed to ensure that appropriate procedures existed to prevent conflicting activities which led to the opening of the pressurizer power operated relief valves (PORVs) when plant conditions did not require them to be open, leading to a partial plant depressurization during plant heat-up. Entergy entered this issue into their corrective action program as CR-IP3-2007-01691. Entergy took immediate corrective action to stop the reactor coolant system pressure transient, and they generated corrective actions to clarify the applicable procedure pre-requisites.

The inspectors determined that this finding was more than minor because it was associated with the Procedure Quality attribute of the Initiating Events cornerstone; and, it affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the lack of procedure clarity and poor interpretation of a procedure pre-requisite led to a loss of reactor coolant system pressure as a result of the pressurizer PORV actuation. This finding was evaluated using Phase 1 of IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At Power Situations." The inspectors determined that the finding was of very low safety significance because assuming the worst case degradation, the loss of inventory did not exceed the Technical Specification limit for identified reactor coolant system (RCS) leakage, and the finding would not have caused a total loss of another mitigating system safety function.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because the applicable procedure prerequisites were not adequate as written to prevent a plant transient. (Section 1R20)

Cornerstone: Mitigating Systems

Green. The inspectors identified a Green, non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," because Entergy failed to generate a procedure of a type appropriate to the circumstances associated with the implementation of a temporary modification to normal control room lighting power. The procedure that was generated lacked precautions, limitations, and prerequisites to prevent a low lighting condition in the control room from existing during implementation of the temporary modification. Consequently, during implementation of this temporary modification there were several control panels that did not have adequate lighting for operators to conduct control board manipulations. Entergy entered this issue into the corrective action program as CR-IP3-2007-00821, took immediate corrective action to add additional lighting to the control room, and generated a contingency procedure to allow backup lighting to be energized, if needed.

The inspectors determined that this finding was more than minor because it caused an actual condition to exist in the control room where lighting at selected control panels was not adequate, and contingency plans were not developed for the potential cases where the temporary lighting that was provided could be lost. This condition was similar to IMC 0612, Appendix E, Example 4.d. Specifically, the lowered level of lighting in the control room was determined to significantly impact the operator's ability to perform certain tasks. The inspectors determined that this finding was not suitable for evaluation using the significance determination process. Consequently, it was reviewed by NRC management and determined to be a finding of very low safety significance in accordance with NRC IMC 0609 Appendix M, "Significance Determination Process Using Qualitative Criteria," because the condition existed for a very limited period of time, other contingency lighting would have been available to the control room staff, and the approximated risk as determined by the regional NRC Senior Reactor Analyst was determined to be very low.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not provide an adequate procedure to implement a temporary modification, in that it lacked precautions, limitations, and prerequisites that ultimately resulted in degraded control room lighting. (Section 1R23)

Cornerstone: Emergency Preparedness

Green. The inspectors identified a Green finding because Entergy failed to take adequate corrective actions for an issue associated with monitoring of service water intake bay level. This deficiency could have prevented identification of entry conditions for an emergency action level. Entergy entered this issue into the corrective action program as CR IP3-2007-00453, and initiated several corrective actions, including plans for enhanced monitoring of service water bay levels, backwashing of trash racks, procedural upgrades, correction of service water bay level instrumentation modification installation, development of modifications for enhanced service water level monitoring equipment, and enhanced inspection and cleaning of intake structure trash racks.

The inspectors determined that this finding was more than minor because it was associated with the Emergency Preparedness cornerstone attribute of facilities and equipment; and, it affected the cornerstone objective of ensuring that a licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Specifically, inadequate monitoring of service water intake bay level could have resulted in failure to declare a notification of unusual event (UE). The inspectors reviewed the EAL entry criteria and determined that this performance deficiency did not affect Entergy's ability to declare any event higher than a UE. The inspectors evaluated this finding using IMC 0609, Appendix B, "Emergency Preparedness Significance Determination Process," Sheet 1, "Failure to Comply," and determined that it was of very low safety significance because the declaration of a UE based on low service water bay level could have been missed or delayed, consistent with the example provided in the appendix.

The inspectors determined that this finding had a cross-cutting aspect in the area of problem identification and resolution because Entergy did not implement effective corrective actions for a previously identified issue associated with inadequate monitoring of service water intake bay level. (Section 1R17)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

Indian Point Nuclear Generating Unit 3 operated at or near full power throughout the inspection period until the beginning of refueling outage 3R14 on March 7, 2007. Entergy conducted a plant startup and synchronized to the electrical grid on March 31, 2007.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 2 samples)

a. Inspection Scope

For the onset of cold weather conditions, the inspectors reviewed the readiness for extreme weather conditions of risk-significant systems. The inspectors reviewed Entergy's adverse weather procedures, operating experience, corrective action program (CAP), Updated Final Safety Analysis Report (UFSAR), Technical Specifications (TS), operating procedures, staffing, and applicable plant documents to determine the types of adverse weather challenges to which the site is susceptible.

The inspectors performed plant walkdowns and reviews to verify that plant features and procedures for operation and continued availability of the ultimate heat sink during adverse weather were appropriate including equipment availability for performance of the reactor shutdown function under the weather conditions assumed prior to shutdown. The documents reviewed during this inspection are listed in the Attachment. The service water, backup service water and the circulating water systems are risk-significant systems that were required to be protected from adverse weather conditions and were selected for inspection. Collectively this inspection represented one inspection sample of risk-significant systems.

Additionally, the inspectors evaluated Entergy's implementation of the adverse weather preparation procedures and compensatory measures for the affected conditions before the onset of and during adverse weather conditions. Specifically, the inspectors reviewed preparations by Entergy for a significant snow storm that occurred on February 12, 2007. This inspection of this severe weather event constituted one inspection sample for onset of severe weather.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Walkdown (71111.04Q - 3 samples)

a. Inspection Scope

The inspectors performed three partial system walkdowns to verify the operability of redundant or diverse trains and components during periods of system train unavailability or following periods of maintenance. The inspectors referenced the system procedures, the UFSAR, and system drawings to verify that the alignment of the available train was proper to support its required safety functions. The inspectors also reviewed applicable condition reports and work orders to ensure that Entergy had identified and properly addressed equipment discrepancies that could potentially impair the capability of the available train. The documents reviewed during this inspection are listed in the Attachment. The inspectors performed partial walkdowns of the following systems, which represented three inspection samples:

- 32 emergency diesel generator (EDG) and 33 EDG during 31 EDG maintenance activities;
- 31 and 32 residual heat removal pumps following testing activities; and
- Component cooling water (CCW) pump and heat exchanger alignment during 32 CCW heat exchanger associated repairs.

b. Findings

No findings of significance were identified.

.2 Complete Walkdown (71111.04S - 1 sample)

a. Inspection Scope

The inspectors performed a complete walkdown of accessible portions of the residual heat removal (RHR) system to identify any discrepancies between the existing equipment lineup and the required lineup. The inspectors reviewed operating procedures, surveillance tests, piping and instrumentation drawings, equipment lineup check-off lists, and the UFSAR to determine if the system was aligned to perform its required safety functions. The inspectors reviewed a sample of condition reports and work orders written for deficiencies associated with the RHR system to ensure that they had been appropriately evaluated and resolved. The documents reviewed during this inspection are listed in the Attachment. The walkdown of the RHR system represented one inspection sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q - 8 samples)

a. Inspection Scope

The inspectors conducted tours of the eight areas listed below to assess the material condition and operational status of fire protection features. The inspectors verified that combustibles and ignition sources were controlled in accordance with Entergy's administrative procedures; fire detection and suppression equipment was available for use; passive fire barriers were maintained; and compensatory measures for out-of-service, degraded, or inoperable fire protection equipment were implemented in accordance with Entergy's fire plan. The inspectors used procedure ENN-DC-161, "Transient Combustible Program," in performing the inspection. The inspectors evaluated the fire protection program against the requirements of License Condition 2.H. The documents reviewed during this inspection are listed in the Attachment. This inspection satisfied eight inspection samples of fire protection tours. The areas inspected included:

- Fire Zones 10, 36A, 101A, 102A;
- Fire Zones 2, 2A;
- Fire Zones 23, 52A;
- Fire Zone 17A, 19A, 20A, 21A;
- Fire Zone 385;
- Fire Zone 86A;
- Fire Zones 4A, 6A, 9; and
- Fire Zones 7A, 60A, 73A, 74A.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 1 sample)

a. Inspection Scope

The inspectors reviewed selected risk-significant plant design features and Entergy's procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed flood analysis and design documents, including the Individual Plant Examination (IPE) and the UFSAR, engineering calculations, and abnormal operating procedures. In addition, the inspectors reviewed three areas and associated equipment which may be affected by internal flooding of the auxiliary feedwater (AFW) building. The areas inspected were flood zones AFW 43, AFW 18-1, and AFW 18-2. The documents reviewed during this inspection are listed in the Attachment. This inspection represented one inspection sample of internal flood protection.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07A - 1 sample)

a. Inspection Scope

The inspectors performed an inspection of the 31 emergency diesel generator (EDG) jacket water cooler. The inspectors verified that Entergy used the periodic maintenance method outlined in Electric Power Research Institute document NP-7552, "Heat Exchanger Performance Monitoring Guidelines." The inspectors reviewed the results of the last inspection conducted on May 9, 2006, and eddy current testing conducted on June 7, 2005, for the jacket water cooler and observed portions of the inspection conducted on March 21, 2007 under work order IP3-06-15718. The documents reviewed during this inspection are listed in the Attachment. The inspection of 31 EDG jacket water cooler represented one inspection sample.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (71111.08 - 7 Samples)

a. Inspection Scope

The inspection assessed the effectiveness of Entergy's program for monitoring degradation of the reactor coolant system boundary. The inspection focused on the boric acid corrosion control and nondestructive examination activities on Class 1 and 2 piping as well as containment system boundaries. The four steam generators were eddy current tested during this outage (3R14) and 32 steam generator had several of its shell welds examined by ultrasonic testing (UT).

For the nondestructive examination activities, the inspectors conducted interviews with the UT, radiographic (RT) and visual (VT) examination personnel and engineering personnel to assess the planning, preparation and conduct of the activities. The inspectors reviewed training and qualification records to verify Entergy's personnel qualification process adequately prepared the assigned staff to perform the examinations. The examination procedures were reviewed to determine whether they provided adequate guidance and examination criteria to implement the examination plan. Prior to a manual UT on the steam generator lower head inner radius and the re-examination of an indication in the upper reactor head to flange weld, the inspectors witnessed the calibration of the ultrasonic equipment. In addition, the inspectors observed a demonstration of the procedure to verify that the calibrated UT equipment would be able to find and accurately characterize flaws on the examined welds. The inspectors observed the subsequent performance and documentation of UT results in the field for the inner radius exam.

During the detection phase by UT of shell weld number 6 in the 32 steam generator, a number of indications were identified and documented in CR-IP3-2007-01456. The inspectors reviewed the results of the UT detection and the subsequent UT sizing, characterization and evaluation of the indications.

Records of an indication left in service in a section of the reactor head-to-flange weld were reviewed to ascertain whether the flaw had changed in magnitude since it was first examined.

The inspectors reviewed the VT examination results of four hot leg and four cold leg safe end to reactor vessel welds that were of interest per the evaluation guideline as inconel dissimilar metal welds. The video records of the examination were in accordance with station procedures and did not show any leakage associated with the dissimilar metal welds.

In the area of radiography, the inspectors reviewed the radiographs for a replacement to the steam generator blowdown line welded in October 2006, report 06R037, and those illustrative of the technique for assessing service water pipe degradation by raw water corrosion. The inspectors assessed the ability of Entergy's inspection activities to identify boric acid corrosion and leaks. Entergy's boric acid inspection procedure was reviewed to determine if it provided adequate scope and guidance on examination criteria and corrective action required when boric acid deposits are found. The inspectors conducted a boric acid walkdown of containment to verify that plant staff effectively inspected for active boric acid leaks. The inspectors reviewed Entergy's boric acid walkdown report for indications of active boric acid leaks or boric acid corrosion of carbon steel components, associated condition reports and corrective actions assigned.

<u>Reactor Pressure Vessel Head Penetration Nozzles- Control Rod Drive Mechanisms</u>

The inspection assessed the effectiveness of Entergy's reactor pressure vessel (RPV) and vessel head penetration (VHP) nozzle inspection in detecting small amounts of boric acid, primary water stress corrosion cracking (PWSCC) in VHP nozzles, and boric acid flow through the interference zone of the fit of the VHP nozzles. The inspection consisted of interviews with UT, eddy current testing (ECT), and VT personnel, data analysts, and engineering personnel. The data analysts' training and qualification records were reviewed to verify that Entergy's personnel qualification process adequately prepared the assigned staff to perform the examination and analysis of accumulated non-destructive examination data. Also, the inspectors reviewed the examination procedures to determine the adequacy of the guidance and examination criteria to implement the examination plan.

For the visual examination of the reactor head bare metal surface, the inspectors observed that the camera operator used the appropriate test chart characters for the VT examination. The inspectors observed the analyst who was reviewing the recorded tapes to verify that the approved procedures were being followed and appropriate examination criteria were available to the analyst and being used to disposition any degraded conditions or evidence of boron. The inspectors verified that appropriate corrective action was taken for indications identified during the examination process, including thorough documentation and effective cleaning of the head and penetrations. The inspectors also verified that Entergy made all efforts to access and inspect the required surface area surrounding the penetrations on top of the reactor head. The inspectors reviewed the video VT records above the upper head for control rod drive mechanisms 10, 14, 15, 16, 21, 22, 23, 24, 26, 30, 31, 38, 39, 41, 65, and 76, to verify that no cracks or significant indications associated with primary water stress corrosion cracking were present in the reactor upper head assembly.

For the ultrasonic/eddy current examination of the RPV penetrations, from under the upper head, the inspectors noted that full coverage of the J-groove weld area was achieved. Portions of the examination were observed to verify that the approved procedures were being followed. The ultrasonic and eddy current records for several CRDM nozzles were accessed and reviewed to confirm the validity of the analyst's assessment of the integrity of the pressure boundary.

Lower Head Penetration (LHP) Nozzles

The inspectors reviewed the LHP nozzle examination procedure to determine whether it provided adequate guidance and examination criteria to implement Entergy's examination plan. The inspectors reviewed examination personnel training and qualification records to ensure that personnel were adequately prepared to perform the assigned examination activities.

The inspectors observed a portion of the LHP inspection activities and also reviewed photographs and examination reports to determine whether the inspection procedure was effectively implemented. The inspectors reviewed the video VT records of the lower head penetrations 10, 13, 14, 19, 22, 26, 34, 41, 48, 53, and 56, to verify that no cracks or significant indications associated with primary water stress corrosion cracking were present in the reactor upper head and to verify that the penetration intersection location could be fully accessed to perform a 360-degree examination.

Steam Generator (SG) Tube Inspection Activities

The inspectors reviewed the Steam Generator Degradation Assessment for the refueling outage 3R14 Engineering Report No. IP-RPT-06-00186, Revision 0, to determine the pre-outage known status of the SG tubes, applicable degradation mechanisms and planned inspection techniques. The extent of 3R14 SG tube ECT examinations were compared to the plan as well as to the results of the ECT examinations. The plan addressed the areas of potential degradation (based on site-specific experience and industry experience) to be inspected, especially areas which are known to represent potential ECT challenges (e.g. top-of-tubesheet, tube support plates, and U-bends). No new degradation mechanisms were identified during the steam generator eddy current testing in 3R14.

In-situ pressure testing was not conducted on steam generator tubing during 3R14 as no tube degradation that would require in-situ pressure testing was identified. Comparison of the estimated size and number of tube flaws detected during the current outage was consistent with the degradation assessment predictions that limited expected tube degradation to various wear mechanisms.

The inspectors confirmed that the SG tube ECT scope and expansion criteria meet Technical Specification (TS) requirements, EPRI Guidelines, and commitments made to the NRC .

The inspectors noted the plugging method applied to the two tubes that were plugged during this outage was approved. The total number of tubes that have been plugged since replacement of the steam generators in 1989 is sixteen tubes. The inspectors verified that the number of plugged tubes is well below the plugging limit of 10 percent.

While the depth sizing repair criterion (typically 40 percent through wall) is being applied for indications other than wear or axial primary water stress corrosion cracking in dented tube support plate intersections, there were no tubes with indications of cracking or approaching the 40 percent limit.

No steam generator tube leakage was identified during the past operating cycle or during post-shutdown visual inspections of the tubesheet face.

The inspectors confirmed that the ECT examination procedures, probes and equipment are qualified for the expected types of tube degradation, and was evaluated by the independent quality data analyst (IQDA) for the scope of the eddy current inspection. The inspectors reviewed the IQDA report dated March 20, 2007 regarding the pre and post outage tasks of IQDA coverage of steam generator eddy current examination for the 3R14 outage.

Where Entergy had identified loose parts or foreign material on the secondary side of the steam generator, the corrective actions included removal of the material, confirmation of the tube integrity and in one case the pre-emptive plugging of two affected tubes. The inspectors confirmed that Entergy had taken appropriate corrective actions for the affected SG tubes, and inspected the secondary side of the SG to remove foreign objects.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Inspection (71111.11Q - 1 sample)

a. Inspection Scope

On January 25, 2007, the inspectors observed licensed operator simulator training to assess operator performance during several scenarios to verify that operator performance was adequate and evaluators were identifying and documenting crew performance problems. The inspectors evaluated the performance of risk significant operator actions, including the use of emergency operating procedures. The inspectors assessed the clarity and effectiveness of communications, the implementation of appropriate actions in response to alarms, the performance of timely control board operation and manipulation, and the oversight and direction provided by the shift manager. The inspectors also reviewed simulator fidelity with respect to the actual plant. Licensed operator training was evaluated against the requirements of 10 CFR 55, “Operators’ Licenses.” The documents reviewed during this inspection are listed in the Attachment. This observation of operator simulator training represented one inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q - 3 samples)

a. Inspection Scope

The inspectors reviewed performance-based problems involving selected structures, systems, or components (SSCs) to assess the effectiveness of the maintenance program. Reviews focused on:

- Proper Maintenance Rule scoping;
- Characterization of reliability issues;
- Changing system and component unavailability;
- 10 CFR 50.65 (a)(1) and (a)(2) classifications;
- Identifying and addressing common cause failures;
- Trending of system flow and temperature values;
- Appropriateness of performance criteria for SSCs classified (a)(2); and
- Adequacy of goals and corrective actions for SSCs classified (a)(1).

The inspectors reviewed system health reports, maintenance backlogs, and Maintenance Rule basis documents. The inspectors evaluated the maintenance program against the requirements of 10 CFR 50.65. The documents reviewed during this inspection are listed in the Attachment. The following maintenance rule samples were reviewed and represented three inspection samples:

- Reactor protection system relays;
- Service water system; and
- Instrument air system.

b. Findings

Introduction. The inspectors identified a Green, non-cited violation (NCV) of 10 CFR 50.65(b), in that, Entergy did not include the Indian Point Unit 3 trash rack structures within the scope of the maintenance rule monitoring program. Additionally, Entergy did not demonstrate the performance or condition of the trash racks was being effectively controlled through the performance of appropriate preventive maintenance such that the structure remained capable of performing its intended function.

Description. On February 5, 2007, Entergy declared a notification of unusual event (UE) condition when Service Water bay levels at Indian Point Unit 3 decreased below the emergency action level threshold value of -4 feet 5 inches mean sea level. The cause of the event was determined to be fouling of the intake structure trash racks such that there was insufficient flow through the trash racks and, when combined with a low tide condition, resulted in service water outlet flow exceeding inlet flow resulting in service water bay levels being drawn down. The low service water bay levels were not identified until the control room operators responded to an “Intake Structure or Traveling Screen

Trouble" alarm, and determined that the screen wash pumps for the traveling water screens tripped on low suction pressure. Entergy closely monitored service water bay levels while developing and implementing action plans to enhance service water bay levels by reducing circulating water flow, backwashing the circulating water system, and obtaining diving services to clear the trash racks.

Following the event, inspectors identified that the intake structure trash racks were not included in Entergy's maintenance rule monitoring program and that the performance or condition of the trash rack structures was not being effectively controlled through the performance of an appropriate preventive maintenance program. Over the past several years, Entergy had established recurring preventive maintenance cleaning schedules of the trash racks that ranged initially from every refueling outage, to the current periodicity of every four years. The next cleaning was not originally scheduled to be performed until November 2007, but has recently been performed due to the February 2007 event. The inspectors identified that several opportunities existed for Entergy to identify and address the lack of a formal inspection plan for the trash racks, including a debris-clogging event in December 2005 that resulted in a non-cited violation in the area of emergency planning. Specifically, Entergy performed emergent cleaning because debris fouling had adversely impacted circulating water and service water bay levels, and service water bay levels had been closely approaching the emergency action level threshold value under the Emergency Plan. Corrective actions from this 2005 issue focused on inadequacies in monitoring of service water bay levels (See Section 1R17.2) and failed to address the adequacy of preventive maintenance cleaning of the trash racks.

The inspectors determined that the intake structure trash racks should be within the scope of Entergy's maintenance rule program because, per 10CFR50.65(b)(2)(ii), failure could prevent the safety-related service water system from fulfilling its safety-related function, and, per 10CFR50.65(b)(2)(iii) failure could cause a reactor scram due to the loss of circulating water.

Entergy performed a cleaning of the trash racks to immediately address the lowered service water intake bay level, and they timed service water bay level monitoring to coincide with river low tide cycles. Entergy also entered this issue into the corrective action program as CR-IP3-2007-00453, and developed corrective actions to: modify the requirements for inspection and cleaning of trash racks based on component history and condition monitoring; modify guidance for service water bay level monitoring to be more effective; evaluate maintenance rule system scoping; develop procedural guidance for managing low service water bay levels; and implement a method for monitoring debris fouling of the trash racks.

<u>Analysis</u>. The inspectors determined that Entergy's failure to include the intake structure trash racks within the scope of their maintenance rule program was a performance deficiency and did not meet the requirements of 10 CFR 50.65(b) which specifies, in part, that the scope of the monitoring program shall include non-safety related structures, systems, or components (SSC's) whose failure could prevent safety-related SSCs from fulfilling their safety-related function, or, whose failure could cause a reactor scram. Traditional enforcement does not apply because there were no actual safety

consequences or potential for impacting the NRC's regulatory function, and the finding was not the result of any willful violation of NRC requirements or Entergy's procedures.

The inspectors determined that this finding affected the Initiating Events cornerstone and was more than minor because it was similar to Example 7.d in IMC 0612, Appendix E, "Examples of Minor Issues." Specifically, equipment performance problems were such that Entergy was unable to demonstrate effective control of the performance or condition of the trash racks through appropriate preventive maintenance as specified by 10CFR50.65(a)(2). The inspectors evaluated the significance of this finding using Phase 1 of IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that the finding was of very low safety significance (Green) because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available.

Enforcement. 10 CFR 50.65 (b) specifies, in part, that the scope of the monitoring program specified in paragraph (a)(1) shall include non-safety related structures, systems, or components whose failure could prevent safety-related SSCs from fulfilling their safety-related function, or, whose failure could cause a reactor scram. 10 CFR 50.65 (a)(2) states, in part, that monitoring as specified in 10 CFR 50.65 (a)(1) is not required where it has been demonstrated that the performance or condition of an SSC is being effectively controlled through the performance of appropriate preventive maintenance, such that the SSC remains capable of performing its intended function. Contrary to the above, following Entergy's declaration of a notification of unusual event due to service water bay levels decreasing below emergency action level thresholds on February 5, 2007, it was identified that Entergy had not included the intake structure trash racks within the scope of their maintenance rule program and failed to demonstrate their performance or condition had been effectively controlled through the performance of appropriate preventive maintenance. Specifically, the intake structure trash racks had been noted to be fouled on several occasions (twice adversely impacting intake bay levels), well short of the planned four-year maintenance periodicity, which demonstrated that the performance was not being effectively controlled through appropriate preventive maintenance. Because this issue is of very low safety significance and is entered into the Entergy's corrective action program, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: **(NCV 05000286/2007002-01, Failure to include the intake structure trash racks within the scope of the maintenance rule monitoring program)**

1R13 Maintenance Risk Assessment and Emergent Work Control (71111.13 - 5 samples)

a. Inspection Scope

The inspectors reviewed planned or emergent activities to verify that the appropriate risk assessments were performed prior to removing equipment from service for planned work. The inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4), and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The documents reviewed during this inspection are listed in the Attachment.

The following three emergent activities and two planned activities were observed and treated as five inspection samples:

- Work order (WO) IP3-05-17211, test of Appendix 'R' alternate feeds to 31 and 32 charging pumps;
- WO IP3-05-14143, main steam safety valve setting verification;
- CR-IP3-07-00501, Entry into abnormal operating procedure for fuel handling due to slipped fuel;
- Compensatory measures for low service water bay level; and
- Buchanan switch yard emergent outages.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 5 samples)

a. Inspection Scope

The inspectors reviewed operability determinations to assess the acceptability of the evaluations, the use and control of compensatory measures, and compliance with Technical Specifications. The inspectors' review included a verification that the operability determinations were made as specified by ENN-OP-104, "Operability Determinations." The technical adequacy of the determinations was reviewed and compared to the TS, UFSAR, and associated design basis documents. The documents reviewed during this inspection are listed in the Attachment. The following evaluations were reviewed and represented five inspection samples:

- CR IP3-2006-03763, service water leak upstream of SWN 34-1;
- CR IP3-2007-00125 & CR IP3-2007-00131, steam generator level indicator isolators LM-437A, LM-437C and LM-447A "as found" data out of specification low at high end of band during missed surveillance;
- CR IP3-2007-00399, residual heat removal (RHR) gas void downstream of RHR valve AC-732;
- CR IP3-2006-03383, ultra low sulphur EDG fuel; and
- CR IP3-2007-01432, reactor vessel specimen sample cap damage.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17A - 2 samples)

Annual Inspection

a. Inspection Scope

The inspectors reviewed modification documents and reviewed the installation and testing of modifications to the Indian Point Nuclear Generating Unit 3 service water bay in accordance with modification ER-05-25451, "Mounting of Permanent Service Water Bay Level Indication." The modifications added level indicators to the Indian Point Unit 2 and Indian Point Unit 3 service water bay to provide low water level indications in support of Emergency Action Level criteria. The modification to install a post with calibrated level markings was completed under work order IP3-05-25367.

The inspectors also reviewed a modification associated with upgrades to the vapor containment and recirculation sumps. This modification was implemented using ER-06-3-005, "IP3 Emergency Core Cooling System Sump Strainer Upgrade," to address concerns associated with pressurized-water reactor containment sump clogging. The inspectors reviewed the modification package to ensure it was technically adequate and conducted walkdowns of the modification to verify it was completed in accordance with the design. The inspectors reviewed quality control records to verify the strainers were installed correctly, interviewed Entergy quality control personnel, and observed Entergy inspectors performing post-installation strainer inspections. The inspectors reviewed in-progress engineering changes to ensure they would not have an adverse effect on sump operability. The inspectors evaluated their observations against the requirements of 10 CFR Part 50.59, "Changes, Tests, and Experiments;" 10 CFR Part 50, Appendix B; and Technical Specifications. The documents reviewed during this inspection are listed in the Attachment. The review of these modifications represented two inspection samples.

b. Findings

.1 Introduction. A Green, self-revealing, non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified, in that, Entergy's work package failed to ensure that piping interference was correctly planned for and removed during modifications to the vapor containment and recirculation sumps.

Description. During pre-outage walkdowns for removal of piping interference associated with replacement of the recirculation pumps, a section of piping (a 1 inch containment spray header drain line to the reactor coolant drain tank (RCDT), line #797) was incorrectly identified and marked for removal. Specifically, the work order package was developed with the wrong section of piping identified for removal as the 32 RHR heat exchanger discharge relief (SI-733A) drain line.

On March 8, 2007, a prejob brief was held to discuss the removal of the piping interference associated with the modification, and on March 9, 2007, line #797 was physically cut from the system. During the actual piping removal, the inspectors noted that maintenance personnel failed to utilize the work order package, which included the specific instructions described below:

- “Prior to breaching of any system (cutting into or opening) a double verification shall be performed to ensure that work is being performed on the correct system/piping/component;”
- “Verify system has been tagged out (mechanically and electrically) and is acceptable to begin work;”
- “Break discharge flange on relief valve 733A using sparkless tools;”
- “Contact Chemistry to verify no hydrogen present prior to removal of SI-733A discharge line to the pressurizer relief tank;” and
- “Verify adequate supports between cut line and relief valve 733A.”

The step that required double verification to ensure work was performed on the correct piping was signed off as complete, because workers had noted the pre-existing markings that indicated which section of piping was scheduled for removal. Additionally, the markings had confirmed for the workers that the field conditions at the job site were as briefed. The inspectors also noted that the remaining steps noted above were left blank.

On March 11, 2007, operators attempted to configure the plant for reactor cavity cleanup following the removal of the reactor vessel head, and raising the water level in the reactor cavity. This cavity cleanup configuration resulted in the formation of a drain path between the reactor coolant loops and the recirculation sump work site (via the RCDT) since there were no isolations in place for the piping section #797, which had been removed two days earlier. Workers observed water entering the recirculation sump from the cut piping. Operations personnel were notified and the leakage path was isolated. Approximately 385 - 500 gallons of reactor coolant water was inadvertently drained from the reactor cavity to the recirculation sump, and two workers received minor radioactive contamination on their garments.

Entergy's immediate corrective actions included generation of a revision to the work order that subsequently welded a cap on the open piping leading from the reactor coolant drain tank to the work site, and placing tags on the residual heat removal interface valves (SI-864E and 864F) to isolate the work area. Entergy entered this issue into the corrective action program as CR-IP3-2007-01059, performed a root cause analysis, and conducted a human performance error review.

Analysis. The inspectors determined that Entergy’s failure to develop an adequate procedure for removal of recirculation pump interference piping and failure to accomplish work in accordance with procedures did not meet the requirements of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” and was a performance deficiency which was reasonably within Entergy’s ability to foresee and prevent. Traditional enforcement does not apply since there were no actual safety consequences or potential for impacting the NRC’s regulatory function, and the finding was not the result of any willful violation of NRC requirements or Entergy procedures.

The inspectors determined that this finding was more than minor because it was associated with the Procedure Quality attribute of the Initiating Events cornerstone; and, it affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, cutting the wrong pipe resulted in the inadvertent draining of reactor coolant system inventory and increased the likelihood of a loss of inventory control. This finding was evaluated using Phase 1 of IMC 0609, Appendix G, “Shutdown Operations Significance Determination Process.” The inspectors evaluated the plant conditions (cold shutdown, reactor coolant system open, refueling cavity less than 23 feet) in accordance with Checklist 3 of Appendix G, Attachment 1, and determined that the finding was of very low safety significance because it did not satisfy the criteria of Table 1 for a "Loss of Control," and the Checklist 3 criteria for maintaining adequate mitigation capability (Core Heat Removal Guidelines, Inventory Control Guidelines, Power Availability Guidelines, Containment Control Guidelines, and Reactivity Guidelines) were met.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because the work package used for interference removal was not accurate and did not ensure the correct section of piping was identified and appropriately controlled.

Enforcement. 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions. Contrary to the above, prior to March 9, 2007, Entergy did not develop an appropriate work package for removal of piping interference for the recirculation sump modification. Entergy entered this issue into their corrective action program (CAP) as (CR-IP3-2007-01059), and they conducted a root cause analysis and a human error prevention review. Because this issue was of very low safety significance and was entered into Entergy’s CAP, this violation is being treated as an NCV per Section VI.A of the NRC Enforcement Policy: **(NCV 05000286/2007002-02, Inadequate Procedure for Recirculation Sump Interference Removal.)**

.2 Introduction. A Green, self-revealing, finding was identified because Entergy failed to take adequate corrective actions for an issue associated with monitoring of service water intake bay level. Specifically, Entergy’s daily performance of intake bay level measurements could have prevented identification of entry conditions for an emergency action level (EAL) under the Emergency Plan.

Description. In November 2005, NRC inspectors identified a Green NCV because Entergy did not have adequate indications available to determine if the entry condition for a notification of unusual event (UE) had been met. Specifically, EAL 8.4.3 requires declaration of a UE if service water intake bay level reaches 4 feet 5 inches below mean sea level. At the time, Entergy did not have an established means to measure intake bay level, or any instrumentation available to plant operators to assess intake bay level, as required by 10 CFR 50.47(b)(4). The NRC issued NCV 05000247/2005005-05, “Inadequate Equipment to Assess Threshold for Emergency Action Level 8.4.3.” In response, Entergy entered the issue into the corrective action program and installed a level measuring device in the service water intake bay.

On February 5, 2007, Indian Point Units 2 and 3 experienced low levels in the service water intake bay due to a combination of debris clogging of the intake trash racks and an unusually low tide. Operators were alerted to this condition because the Indian Point 3 non-safety-related screen wash pumps had tripped due to low suction pressure, resulting in a control room alarm. Indian Point Unit 3 operators responded to the intake bay area, observed the installed, intake bay level measuring device, and determined that the entry conditions for a UE were met. Indian Point Unit 3 operators declared a UE at 7:07 a.m. on February 5, which was terminated at 10:14 a.m. when water level increased above the UE entry conditions. Indian Point Unit 2 also experienced lower than normal service water intake bay levels, but did not meet the entry conditions for a UE.

Following the February 2007 UE, the inspectors reviewed Entergy's corrective actions from the November 2005 NCV. The inspectors reviewed Entergy's method of monitoring service water intake bay level, and reviewed alarm response and abnormal operating procedures associated with service water system. The inspectors determined that while Entergy had installed a measuring device, it was not used in a manner to provide assurance that the entry conditions for a UE would be identified in a timely manner. Specifically, while the device was used to measure intake level as a part of operator rounds, the readings were not trended and were only recorded once per day with no time specified for when intake bay level should be measured. As a result, the readings could potentially be taken during periods of high tide, which could mask subsequent low level conditions in the service water intake bay. Additionally, the inspectors reviewed both alarm response procedures and abnormal operating procedures, and identified that existing plant procedures did not provide sufficient guidance to operators to identify and mitigate low level conditions in the intake bay. Plant procedures did not direct the operators to check service water intake bay level following the trip of screen wash pumps, required no specific actions if service water bay level was low out of specification on operator logs, and provided no actions to assist operators in mitigating a low level condition, once identified. These issues were also identified by Entergy during their root cause investigation of the February 2007 UE.

Entergy procedure EN-LI-102, "Corrective Action Process," requires that corrective actions address the cause or resolve the deficiency associated with an adverse condition. Attachment 9.2 of EN-LI-102 provides examples of adverse conditions, and includes actual or potential NRC violations, as well as conditions which could negatively impact reliability or availability. The inspectors determined that Entergy's actions to address the previous NCV did not appropriately correct a condition adverse to quality, as required by EN-LI-102.

<u>Analysis</u>. The inspectors determined that Entergy's failure to take adequate corrective actions for the improper monitoring of service water intake bay level was a performance deficiency. This issue was reasonably within Entergy's ability to foresee and prevent, given that the issue had been identified and documented in a condition report and the corrective action requirements were addressed in Entergy procedure EN-LI-102. Traditional enforcement does not apply since there were no actual safety consequences or potential for impacting the NRC's regulatory function, and the finding was not the result of any willful violation of NRC requirements or Entergy procedures.

The inspectors determined that this finding was more than minor because it was associated with the facilities and equipment attribute of the Emergency Preparedness cornerstone; and, it affected the cornerstone objective of ensuring that a licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Specifically, inadequate monitoring of service water intake bay level could have resulted in failure to declare a UE. The inspectors reviewed the EAL entry criteria and determined that this performance deficiency did not affect Entergy's ability to declare any event higher than a UE. The inspectors evaluated this finding using IMC 0609, Appendix B, "Emergency Preparedness Significance Determination Process," Sheet 1, "Failure to Comply." Section 4.4 of IMC 0609, Appendix B, provides examples for use in assessing emergency preparedness findings. One example of a Green finding states, "The EAL classification process would not declare any alert or notification of unusual event that should be declared." Since the declaration of a UE based on low service water bay level could have been missed or delayed, this finding was considered consistent with the example provided and was therefore determined to be of very low safety significance (Green).

The inspectors determined that this finding had a cross-cutting aspect in the area of problem identification and resolution because Entergy did not implement effective corrective actions for a previously identified issue associated with inadequate monitoring of service water intake bay level.

Enforcement. Because this finding is associated with a non-safety-related service water intake bay level monitoring function, no violation of regulatory requirements occurred. Entergy entered this issue into the corrective action procedure as CR IP3-2007-00453, and initiated several corrective actions, including plans for enhanced monitoring of service water bay levels, backwashing of trash racks, procedural upgrades, correction of service water bay level instrumentation modification installation, development of modifications for enhanced service water level monitoring equipment, and enhanced inspection and cleaning of intake structure trash racks. **(FIN 05000286/2007002-03, Inadequate Corrective Actions for Failure to Appropriately Monitor Service Water Intake Bay Level)**

1R19 Post-Maintenance Testing (71111.19 - 5 samples)

a. Inspection Scope

The inspectors reviewed post maintenance test procedures and associated testing activities for selected risk-significant mitigating systems to assess whether the effect of maintenance on plant systems was adequately addressed by control room and engineering personnel. The inspectors verified that test acceptance criteria were clear, demonstrated operational readiness and were consistent with design basis documentation; test instrumentation had current calibrations and the range and accuracy for the application; and tests were performed, as written, with applicable prerequisites satisfied. Upon completion, the inspectors verified that equipment was returned to the proper alignment necessary to perform its safety function. Post-maintenance testing was evaluated against the requirements of 10 CFR 50, Appendix B, Criterion XI, "Test Control." The documents reviewed during this inspection are listed in the Attachment.

The following post-maintenance test activities were reviewed and represented five inspection program samples:

- WO IP3-06-12102, reactor protection logic functional test;
- WO IP3-05-16245, 31 EDG capacity test following 8 year PM;
- WO IP3-03-15984, reactor head vents after reactor reassembly;
- WO-IP3-05-15768, recirculation pumps following pump replacement; and
- WO-IP3-06-23822, replacement of 18 inch piping at inlet of 32 component cooling water heat exchanger.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20 - 1 sample)

a. Inspection Scope

The inspectors reviewed the schedule and risk assessment documents associated with the Indian Point Unit 3 refueling outage 3R14, to confirm that Entergy appropriately considered risk, industry operating experience, and previous site-specific problems in developing and implementing a plan that ensured a maintenance of defense-in-depth. Prior to the refueling outage, the inspectors reviewed Entergy's outage risk assessment to identify risk-significant equipment configurations and to determine whether planned risk management actions were adequate.

The inspectors observed the Unit 3 shutdown and cooldown on March 7, 2007, to verify that cooldown rates met TS requirements. Inspectors also evaluated conditions within containment for indications of unidentified leakage and damaged equipment. The inspectors verified that Entergy managed the outage risk commensurate with the outage plan. Inspectors periodically observed refueling activities from the refueling bridge in containment and the spent fuel pool (SFP) to verify refueling gates and seals were properly installed and to determine whether foreign material exclusion boundaries were established around the reactor cavity. Core offload and reload activities were periodically observed from the control room and refueling bridge to verify whether operators adequately controlled fuel movements in accordance with procedures.

The inspectors verified that tagged equipment was properly controlled and equipment configured to safely support maintenance work. Equipment work areas were periodically observed to determine whether foreign material exclusion boundaries were adequate. During control room tours, the inspectors verified that operators maintained adequate reactor coolant system level and temperature and that indications were within the expected range for the operating mode.

The inspectors determined whether offsite and onsite electrical power sources were maintained in accordance with TS requirements and consistent with the outage risk assessment. Periodic walkdowns of portions of the onsite electrical buses and the emergency diesel generators were conducted during risk-significant electrical configurations. The inspectors verified through routine plant status activities that the decay heat removal safety function was maintained with appropriate redundancy as required by TS and consistent with Entergy's outage risk assessment. During core offload conditions, the inspectors periodically determined whether the spent fuel pool cooling system was performing in accordance with applicable system operating procedures and consistent with Entergy's risk assessment for the refueling outage. Equipment and procedures to mitigate a loss of spent fuel cooling were reviewed by the inspectors to ensure they were available and ready for use.

Reactor coolant system inventory controls and contingency plans were reviewed by the inspectors to determine whether they met TS requirements and provided for adequate inventory control. Inspectors reviewed procedures and observed portions of activities in the control room when the unit was in the reduced inventory mode of operation, including mid-loop operations. Water level and core temperature measurement instrumentation was reviewed by the inspectors to ensure they were installed and operational. Calculations that provided time to core boil information were also reviewed for reactor coolant system reduced inventory conditions as well as for the spent fuel pool during high heat loads.

Containment status and procedural controls were reviewed by the inspectors during fuel offload and reload activities to verify that TS requirements and procedure requirements were met for containment. Specifically, the inspectors verified that during fuel movement activities, personnel, materials, and equipment were staged to close containment penetrations as assumed in the licensing basis.

The inspectors observed plant heat up and start up activities including the approach to criticality. In addition, the inspectors observed the main generator synchronization to the electrical grid, and initial power ascension. The documents reviewed during this inspection are listed in the Attachment. The combined efforts described above represent one inspection program sample.

b. Findings

Introduction. A Green, self-revealing, non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified, in that, Entergy failed to ensure that appropriate procedures existed to prevent conflicting activities which led to the inadvertent opening of the pressurizer power operated relief valves (PORVs), leading to a partial plant depressurization during plant heat-up.

Description. On March 28, 2007, Entergy conducted a plant heat-up as part of the 3R14 refueling outage completion activities. With reactor coolant temperature at 340 degrees Fahrenheit (°F), and reactor coolant pressure at 675 pounds per square inch gauge (psig), both pressurizer PORVs unexpectedly opened, lowering reactor coolant pressure. Because plant conditions did not require the pressurizer PORVs to be open, the Shift Manager ordered the pressurizer PORVs to be manually closed.

The cause of the pressurizer PORV actuation was due to workers incorrectly interpreting a station procedure, as well as a corresponding lack of clarity within this procedure.

Specifically, procedure RA-1, “Incore Thermocouple Wide Range Resistance Temperature Detector (RTD) and Narrow Range RTD Measurement,” Revision 21, contained a prerequisite to ensure that the overpressure protection system (OPS) was “no longer required by procedures.” When the reactor engineer reached this requirement in the procedure, plant conditions (reactor coolant temperature of 319 degrees °F) and TS no longer required OPS to be in service. However, applicable operational procedures would not have deactivated the OPS until a point later in the plant heat up process, when reactor coolant system pressure reached 800 psig. The reactor engineering staff believed that the prerequisite condition was met, and continued with their procedure which removed two wide-range, reactor coolant system cold leg temperature detectors from service. As a result of this action, two of three temperature inputs to the OPS system failed low, which caused the OPS system pressure actuation setpoint to likewise fail low. Actual system pressure was then higher than the OPS system setpoint, and caused the pressurizer PORVs to open.

Analysis. The inspectors determined that Entergy’s failure to develop an adequate procedure for conducting the RTD calibrations did not meet the requirements of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” and was a performance deficiency which was reasonably within Entergy’s ability to foresee and prevent. Traditional enforcement does not apply since there were no actual safety consequences or potential for impacting the NRC’s regulatory function, and the finding was not the result of any willful violation of NRC requirements or Entergy procedures.

The inspectors determined that this finding was more than minor because it was associated with the Procedure Quality attribute of the Initiating Events cornerstone; and, it affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the lack of procedure clarity and poor interpretation of a procedure pre-requisite led to a loss of reactor coolant system pressure due to pressurizer PORV actuation. This finding was evaluated using Phase 1 of IMC 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At Power Situations.” IMC 0609 Appendix G, “Shutdown Operations Significance Determination Process” was not used for the evaluation because plant conditions no longer required operation of the residual heat removal system for decay heat removal. The inspectors determined that the finding was of very low safety significance because assuming the worst case degradation, the loss of inventory did not exceed the Technical Specification limit for identified reactor coolant system leakage, and the finding would not have caused a total loss of another mitigating system safety function.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because the applicable procedure prerequisites were not adequate as written to prevent a plant transient.

Enforcement. 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions. Contrary to the above, on March 28, 2007, Entergy failed to develop a procedure to adequately conduct the incore thermocouple measurements which led to a reactor coolant system pressure transient. Entergy generated corrective actions to clarify the procedure pre-requisite that led to the inadvertent pressurizer PORV actuation. Because this issue was of very low safety significance and was entered into Entergy's corrective action program as CR-IP3-2007-01691, this violation is being treated as an NCV per Section VI.A of the NRC Enforcement Policy: **(NCV 05000286/2007002-04, Inadequate Procedure for Conduct of RTD Cross Calibrations.)**

1R22 Surveillance Testing (71111.22 - 6 samples)

a. Inspection Scope

The inspectors witnessed performance of surveillance tests and/or reviewed test data of selected risk-significant SSCs to assess whether the SSCs satisfied TS, UFSAR, Technical Requirements Manual, and Entergy procedure requirements. The inspectors verified that test acceptance criteria were clear, demonstrated operational readiness and were consistent with design basis documentation; test instrumentation had current calibrations and the range and accuracy for the application; and tests were performed, as written, with applicable prerequisites satisfied. Upon surveillance test completion, the inspectors verified that equipment was returned to the status specified to perform its safety function. The inspectors evaluated the surveillance tests against the requirements in TS. The documents reviewed during this inspection are listed in the Attachment. The following surveillance tests were reviewed and represented six inspection samples (one containment isolation valve in-service test sample, one inservice testing sample, and four surveillance test samples):

- 3PT-Q036, "IST Stroke Test of Valves AC-MOV-822A & B and AC-751A & B (CIV)," Revision 18;
- 3-PC-OL12E, "Containment Pressure Loop P-948C Channel Calibration," Revision 3;
- 3-PT-SA045, "Main Turbine Stop and Control Valves Exercise Test," Revision 2;
- 3-PT-Q116B, "32 SI Pump Surveillance Test (IST)," Revision 13;
- 3-PT-R006A, "Main Steam Safety Valves Setting Test Using Set Pressure Verification Device," Revision 6; and
- 3PT-R160B, "32 EDG Capacity Test," Revision 9.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 2 samples)

a. Inspection Scope

The inspectors reviewed the two temporary modifications listed below. The inspectors assessed the adequacy of the 10 CFR 50.59 evaluations for these temporary modifications including verifying that the installation was consistent with the modification documentation; the drawings and procedures were updated as applicable; and the post-installation testing was adequate. The documents reviewed during this inspection are listed in the Attachment. This inspection satisfied two inspection program samples for temporary modifications.

- TA-04-3-093, temporary repair of a wire related to security equipment, resulting in no security system vulnerability; and
- 3-SYS-018-GEN, procedurally controlled temporary modification to supply control room lighting from MCC 36E.

b. Findings

Introduction. The inspectors identified a Green, non-cited violation of 10 CFR 50 Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” in that, Entergy failed to generate a procedure of a type appropriate to the circumstances associated with the implementation of a temporary modification to normal control room lighting power. The procedure that was generated lacked precautions, limitations, and prerequisites to prevent a low lighting condition in the control room from existing during implementation of the temporary modification.

Description. On March 1, 2007, while at full power, Entergy implemented a procedurally-controlled temporary modification to change the normal power supply for control room lighting from motor control center 36C to motor control center 36E, in preparation for the upcoming outage. This temporary modification was conducted using procedure 3-SYS-018-GEN, “Installation, Control, and Removal of Support Electrical and Mechanical Equipment Required For Scheduled Bus 2A Outages,” Revision 0. In particular, Section 4.4 governs the swapping of power supplies to lighting panel 320, which is the lighting panel which supplies normal control room lighting. During the implementation of this procedure, normal control room lighting was turned off; however, backup direct current lighting did not energize as the operators expected, and the operating crew quickly restored normal control room lighting. It was later determined that this was an expected response based on how power was being removed from lighting panel 320. The operating crew supplied two temporary lights for the control room, and continued with the procedure; however, unknown to the operators, this lighting was energized from wall sockets that would not have been available during certain design basis events. In addition, there were several control panels that did not have adequate lighting to conduct control board manipulations. Procedure 3-SYS-018-GEN did not have operational precautions, limitations, or prerequisites to provide the operating crew with adequate control room lighting or contingency plans during certain operational events because the temporary lighting that was installed would be lost. Once normal control room lighting was restored on the temporary power coming from motor control center 36E, it was realized that there may be a need to manually energize the backup direct current lighting. Therefore, Entergy developed a procedure to direct operators to energize the backup lighting if needed. Entergy entered this issue into their corrective action program as CR-IP3-2007-01691. Entergy took immediate corrective action to stop the reactor coolant system pressure transient, and they plan to clarify the applicable procedure pre-requisites as a part of their corrective actions.

Analysis. The inspectors determined that Entergy's failure to develop an adequate procedure for implementing the control room lighting temporary modification was a performance deficiency. It is reasonable that Entergy should have identified this procedural inadequacy. Traditional enforcement does not apply since there were no actual safety consequences or potential for impacting the NRC's regulatory function, and the finding was not the result of any willful violation of NRC requirements or Entergy's procedures.

The inspectors determined that this finding was more than minor because it caused an actual condition to exist in the control room where lighting at selected control panels was not adequate, and contingency plans were not developed for the potential cases previously described where the temporary lighting that was provided could be lost. This condition was similar to Inspection Manual Chapter 0612, Appendix E, Example 4.d. Specifically, the lowered level of lighting in the control room was determined to significantly impact the operator's ability to perform certain tasks. The inspectors determined that this finding was not suitable for evaluation using the significance determination process. Consequently, it was reviewed by NRC management and determined to be a finding of very low safety significance in accordance with NRC IMC 0609 Appendix M, "Significance Determination Process Using Qualitative Criteria," because the condition existed for a very limited period of time, other contingency lighting would have been available to the control room staff, and the approximated risk as determined by the regional NRC Senior Reactor Analyst was determined to be very low.

The inspectors also determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not provide an adequate procedure to implement a modification, in that, it lacked precautions, limitations, and prerequisites that ultimately resulted in degraded control room lighting.

Enforcement. 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances. Contrary to the above, on March 1, 2007, Entergy did not prescribe a procedure of a type appropriate to the circumstances associated with providing temporary power for control room lighting. Specifically, procedure 3-SYS-018-GEN, "Installation, Control, and Removal of Support Electrical and Mechanical Equipment Required For Scheduled Bus 2A Outages," Revision 0, did not have adequate precautions, limitations, and prerequisites to prevent a low lighting condition in the control room from occurring. Because this violation is of very low safety significance and has been entered into Entergy's corrective action program (CR-IP3-2007-00821), it is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: **(NCV 05000286/2007002-05, Inadequate Procedure for Control of Temporary Modification.)**

Cornerstone: Emergency Preparedness (EP)

1EP2 Alert and Notification System Evaluation (71114.02 - 1 Sample)

a. Inspection Scope

Region-based specialist inspectors evaluated Entergy's corrective actions related to the existing Indian Point alert and notification system (ANS) failures, and reviewed the progress made in the design and installation of the new siren system. Inspection activities were conducted onsite throughout the quarter between January 16 and March 28, 2007. This inspection was conducted in accordance with the baseline inspection program deviation authorized by the NRC Executive Director of Operations (EDO) in a memorandum dated October 31, 2005, and renewed by the EDO in a memorandum dated December 11, 2006.

A new ANS is being installed around the Indian Point Energy Center to satisfy commitments documented in a NRC Confirmatory Order dated January 31, 2006, that implements the requirements outlined in the 2005 Energy Policy Act. In January 2007, Entergy requested an extension of the deadline for completing the ANS project as described in the Confirmatory Order, which set a January 30, 2007, deadline for completion of the installation. Entergy's extension request cited several issues that were beyond their control as the basis for the delay. On January 23, 2007, the NRC granted Entergy's extension request and established April 15, 2007, as the new installation completion date.

The inspectors conducted the following onsite inspection activities during this quarter:

- Assessed Entergy's progress with the new ANS to validate Entergy's justification for the extension of the original Confirmatory Order deadline (January 16, 2007)

- Observed the first full-volume sounding of the new sirens (February 15, 2007)

- Reviewed Entergy's acceptance testing process for transfer of the ANS subsystem components from the vendor to Entergy (February 27-28, 2007)

- Observed and inspected the degraded voltage testing of the back-up batteries for the new ANS as described in the Test Plan for Indian Point Emergency Notification System in accordance with NRC Order EA-05-190 (dated July 5, 2006)

  Note- This testing assured that the batteries at the central control units, the simulcast towers, and the sirens, would operate at their end-of-life condition following a loss of AC power for 24 hours. The inspectors observed the discharge of the batteries at one of the siren locations and at one of the simulcast towers, and observed the subsequent testing of the siren system with the batteries in the degraded condition (March 12-14, 2007).

- Observed and inspected full-volume sounding of the new sirens
  (March 21, 27, and 28, 2007)

During the onsite inspections cited above, the inspectors also reviewed the status of, and corrective actions for, the current ANS to assure that Entergy was appropriately maintaining the system.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06 - 1 sample)

a. Inspection Scope

The inspectors observed an emergency preparedness drill conducted on January 24, 2006. The inspectors used NRC Inspection Procedure 71114.06, "Drill Evaluation," as guidance and criteria for evaluation of the drill. The inspectors observed the drill and critiques that were conducted from the participating facilities on-site, including the Indian Point Unit 2 plant simulator, and the emergency operations facility. The inspectors focused the reviews on the identification of weaknesses and deficiencies in classification and notification timeliness, quality, and accountability of essential personnel during the drill. The inspectors observed Entergy's critique and compared the licensee's self-identified issues with the observations from the inspectors' review to ensure that performance issues were properly identified. The observation of the drill represented one inspection program sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 7 samples)

a. Inspection Scope

On March 19 through 22, 2007, the inspectors conducted the following activities to verify that Entergy was properly implementing physical, engineering, and administrative controls for access to high radiation areas, and other radiologically controlled areas, and that workers were adhering to these controls when working in these areas. Implementation of the access control program was reviewed against the criteria contained in 10 CFR 20, Technical Specifications, and Entergy's procedures.

(1) Radiation work permits were reviewed that provide access to exposure significant areas of the plant including high radiation areas. Specified electronic personal dosimeter alarm set points were reviewed with respect to current radiological condition applicability and workers were queried to verify their understanding of

plant procedures governing alarm response and knowledge of radiological conditions in their work area.

(2) There were no radiation work permits for airborne radioactivity areas with the potential for individual worker internal exposures of >50 mrem committed effective dose equivalent.

(3) Between March 19 through 22, 2007, the following, radiologically-significant work activities were selected; the radiological work activity job requirements were reviewed; and work activity job performance was reviewed with respect to the radiological work requirements:

- Refueling activities;
- Containment sump modification;
- 33 and 34 reactor coolant pump seal replacement activities;
- Reactor cavity drain down and reactor vessel head reinstallation; and
- 31, 32, 33, and 34 steam generator primary manway insert maintenance.

(4) During observation of the work activities listed in (3) above, the adequacy of surveys, job coverage and contamination controls were reviewed.

(5) There were no significant dose gradients requiring relocation of dosimetry for the radiologically significant work activities listed in (3) above.

(6) During observation of the work activities listed in (3) above, radiation worker performance was evaluated with respect to the specific radiation protection work requirements and their knowledge of the radiological conditions in their work areas.

(7) During observation of the work activities listed in (3) above, radiation protection technician work performance was evaluated with respect to their knowledge of the radiological conditions, the specific radiation protection work requirements and radiation protection procedures.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02 - 3 samples)

a. Inspection Scope

During March 19 through 22, 2007, the inspectors conducted the following activities to verify that Entergy was properly maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). Implementation of the ALARA program was reviewed against the criteria contained in 10 CFR 20.1101(b) and Entergy's procedures.

(1) The following highest exposure work activities for the Spring 2007 Unit 3 refueling outage were selected for review:

- Refueling activities;
- Containment sump modification;
- 33 and 34 reactor coolant pump seal replacement activities;
- Reactor cavity drain down and reactor vessel head reinstallation; and
- 31 through 34 steam generator primary manway insert maintenance.

(2) With respect to the work activities listed in (1) above, these job sites were observed to evaluate if surveys and ALARA controls were implemented as planned.

(3) With respect to the work activities listed in (1) above, radiation worker and radiation protection technician performance was observed during the performance of these work activities to demonstrate the ALARA principles.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification (71151- 3 samples)

.1 Mitigating Systems Cornerstone

a. Inspection Scope

The inspectors reviewed performance indicator (PI) data for the below listed cornerstones and used Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 4, to verify individual PI accuracy and completeness.

Initiating Events Cornerstone

- Unplanned scrams per 7000 critical hours
- Unplanned power transients per 7000 critical hours

Barrier Integrity Cornerstone

- RCS activity

The inspectors reviewed data and plant records from December 2005 to December 2006. The records reviewed included PI data summary reports, licensee event reports, operator narrative logs, and maintenance rule records. The inspectors verified the accuracy of the number of critical hours reported, and interviewed the system engineers and operators responsible for data collection and evaluation.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Routine Problem Identification and Resolution (PI&R) Program Review

a. Inspection Scope

As required by Inspection Procedure 71152, “Identification and Resolution of Problems,” and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of all items entered into Entergy’s corrective action program (CAP). The review was accomplished by accessing Entergy’s computerized database for condition reports (CRs) and attending CR screening meetings.

In accordance with the baseline inspection modules, the inspectors selected CAP items across the Initiating Events, Mitigating Systems, and Barrier Integrity cornerstones for additional follow-up and review. The inspectors assessed Entergy’s threshold for problem identification, the adequacy of the cause analyses, extent of condition review, and operability determinations, and the timeliness of the specified corrective actions. The CRs reviewed are listed in the Attachment.

b. Findings and Observations

No findings or observations of significance were identified.

.2 PI&R Annual Sample - Selected Issue Follow-up Inspection - Review of Corrective Actions Associated with Previous Refueling Outage Related NCVs (71152 - 1 Sample)

a. Inspection Scope

The inspectors conducted a review of the effectiveness of corrective actions associated with the four NCVs issued to Indian Point Energy Center for refueling outage related activities. The NCVs reviewed included:

- 50-247/2004-012-06, "failure to follow RCS drain down procedure due to inappropriate approach;"
- 50-247/2006-003-01, "inadequate procedures for placing RHR pump suction pressure gages in service;"
- 50-247/2006-003-06, "failure to provide appropriate procedural direction to ensure that significant volume of gas would not accumulate in the reactor vessel head during shutdown;" and
- 50-247/2006-002-01, "violation of license condition 2.K. for failure to identify a degraded fire door between the 21 and 22 RHR pump cells."

The inspectors interviewed operations personnel responsible for implementing the procedures, reviewed condition reports associated with the violations and documented corrective actions, and assessed Entergy's threshold for problem identification, the adequacy of the causal analysis, and extent of condition review. The documents reviewed during the inspection are listed in the Attachment.

b. Findings and Observations

No findings of significance were identified.

The inspectors observed that Entergy appropriately addressed the causes and extent of condition for these NCVs. Corrective actions for the issues and associated causes were adequate.

.3 PI&R Annual Sample Review: Maintenance Rule Scoping for Emergency Operating Procedure Equipment (71152 - 1 Sample)

a. Inspection Scope

The inspectors conducted a review of CR IP3-2006-00254, which identified that a thorough look at the scoping of SSC's was required to determine Maintenance Rule applicability. This was to ensure that SSC's used in the EOP's were properly scoped following the identification that the control rod drive fans were not within Entergy's Maintenance Rule program as required. The inspectors evaluated the corrective actions associated with this condition report to determine if they were sufficient to correct the identified issue and if the actions taken were effective. The inspectors reviewed the EOP's and cross-referenced to Maintenance Rule SSC's to determine whether any components had been improperly assessed. In addition the inspectors reviewed applicable engineering requests and documentation to support the review.

b. Findings and Observations

The inspectors determined that Entergy's actions to complete this assessment were ongoing. The associated corrective action documents had been closed to an engineering work request that was still open. The control rod drive fans had been included in the Maintenance Rule and developed an appropriate basis document. The inspectors determined that no components were identified that were improperly scoped and would be in an a(1) status based on SSC performance. The inspectors determined that Entergy has made adequate progress on this corrective action, and have adequate plans in place to complete all necessary actions.

4OA3 Event Followup (71153 - 1 sample)

.1 Unusual Event Declaration Based on Low Service Water Bay Levels

a. Inspection Scope

The inspectors observed the response of control room personnel response to a declaration of an Unusual Event on February 5, 2007, due to low service water bay levels. The inspectors observed Entergy's response to verify that plant equipment responded as expected, and to ensure that operating procedures were being appropriately implemented. The inspectors discussed the event and corrective actions with plant management to confirm that Entergy had taken appropriate corrective actions in response to the event. Entergy initiated several corrective actions, including: plans for enhanced monitoring of service water bay levels; backwashing of trash racks; procedural upgrades; correction of service water bay level instrumentation modification installation; development of modifications for enhanced service water level monitoring equipment; and enhanced inspection and cleaning of intake structure trash racks. Entergy also performed a cleaning of the trash racks to immediately address the lowered service water intake bay level.

b. Findings

The inspectors identified two findings associated with the Unusual Event, which are discussed in other sections of this inspection report. The first, a Green non-cited violation of 10 CFR 50.65 (b), scope of the maintenance rule monitoring program, is described in Section 1R12 of this report. The second, a Green self-revealing finding for ineffective corrective actions, is described in Section 1R17 of this report.

.2 (Closed) LER 06000286/2006-002-00 Manual Reactor Trip as a Result of Arcing Under the Main Generator Between Scaffolding and Phase A & B of the Isophase Bus Housing

A manual reactor trip was initiated on July 21, 2006, while the plant was operating at 100 percent power, because plant operators observed electrical arcing between installed scaffolding and the isolated phase bus duct in the vicinity of the main generator. Entergy generated procedural guidance to ensure an adequate distance exists in the future between scaffolding and the isophase bus ducts. The LER was reviewed by the inspectors and no findings of significance were identified, and no violation of NRC requirements occurred. This LER is closed.

4OA5 Other Activities

.1 Groundwater Contamination Investigation

a. Inspection Scope

Continued inspection of Entergy's plans, procedures, and characterization activities regarding the contaminated groundwater condition at Indian Point, relative to NRC regulatory requirements, was authorized by the NRC Executive Director of Operations in a

Reactor Oversight Process deviation memorandum dated October 31, 2005 (ADAMS Accession Number ML053010404) and renewed on December 11, 2006 (ADAMS Accession Number ML063480016). Accordingly, continued oversight of Entergy's progress has been conducted throughout this quarterly inspection period, consisting of on-site inspections; independent split sample analyses of selected monitoring well samples; frequent review of Entergy's performance, progress, and achievements; and periodic communications with Federal, State, and local government stakeholders.

The inspectors conducted an on-site review of tracer test sampling and waterloo sampler maintenance from February 26 to March 2, 2007. A teleconference was held on March 21, 2007, to discuss Entergy's preliminary data and interpretations of their groundwater tracer study, which began on February 8, 2007. The NRC team included representatives from the NRC's Region I office, as well as the NRC's Office of Nuclear Regulatory Research, the U.S. Geological Survey's New York Science Center, and the New York State Department of Environmental Conservation (NYS DEC). The teleconference provided for an independent hydrology review of Entergy's initial tracer test findings and associated re-evaluation of the current site groundwater model.

The tracer test objective uses groundwater tracing techniques by injecting fluorescent tracer dye into a ground location representing the source of leakage and tracks the natural groundwater progress as it is intercepted by existing monitoring wells and storm drain locations. This process better characterizes groundwater flow directions and flow rates in areas identified as being affected by water contaminated with strontium and tritium. The fluoresceine dye was injected into a tracer injection well next to existing monitoring well 30 (MW-30), which is adjacent to the Unit 2 spent fuel pool (SFP). On February 8, 2007, the test began with injection of approximately 200 gallons of dye at a three gallons per minute at a ground elevation equivalent to the bottom of the Unit 2 SFP. The natural groundwater flow of this tracer test is expected to be tracked for approximately 13 weeks by measuring the dye content in charcoal and water samples taken at selected, on-site monitoring wells and storm drain locations.

Initial results indicated that dye tracer was detected within four hours of injection at shallow sampling levels of MW-31 and MW-32. After one day, tracer was detected at deeper levels within MW-31 and in recovery well 1 (RW-1). Direct water sampling was conducted in surrounding wells with carbon sampling devices in outer wells. Once the fluoresceine dye was detected in the carbon sampling devices, direct water sampling was performed to determine the dye concentration. Arrival times and concentrations of the dye were identified in the down-gradient wells and storm drains [e.g., manholes (MH-5 and later MH-6)] as the tracer test progressed. Ozark Underground Laboratory is analyzing the tracer samples and will be reporting their results to Entergy.

b. Findings and Observations

No findings of significance were identified.

The NRC samples were analyzed by the NRC's contract laboratory, the Oak Ridge Institute for Science and Education, Environmental Site Survey and Assessment Program (ORISE/ESSAP) radioanalytical laboratory. The NRC's assessment of Entergy's sample analytical results data generally indicated that their analytical contractor continued to report sample results that were consistent with NRC's analytical results.

The NRC's ORISE/ESSAP sample results are available in ADAMS under the following Accession Numbers: ML070940618, ML070940504, ML070940515, ML070940534, ML070940546, and ML070940574. To date, sample results from site boundary wells and off-site environmental groundwater sampling locations have not indicated any detectable plant-related radioactivity.

NRC's assessment of Entergy's interim tracer test results from February 8 to March 9, 2007, which included input from NYS DEC and U.S. Geological Survey hydrology experts, indicated that an additional complexity to the site groundwater model has been observed with some preferential fracture flow observed in the unsaturated zone (above the water table), as well as a general groundwater flow towards the Hudson River. Additional information will be obtained as the 13 week tracer test progresses to help clarify these initial observations in a later NRC review. Ultimately, clarification of groundwater flow rates of contaminants off-site toward the Hudson River is the focus of this NRC hydrology assessment. Together with monitoring well sample data, an accurate assessment of Entergy's effluent release reports and public dose assessments will result from these efforts.

Entergy and their contractors pointed to the preliminary nature of their data and interpretation. They agreed to provide timely data transfer with a technical meeting in May to review all of the tracer data, arrival times and concentrations. No further pumping in RW-1 or other tracer tests will occur until the data has been reviewed and analyses have been conducted.

Remaining activities identified include: (1) completion of the direct sampling of the tracer in the monitoring wells; (2) preparation and analysis of breakthrough curves (tracer clearance rates) for the tracer at the monitoring wells differentiated by depth; (3) analysis of the breakthrough curve "tails" to determine the nature of groundwater flow (i.e., fracture flow or porous media flow); and (4) correlation of the earlier RW-1 pump test data with the tracer test data to further clarify and corroborate the groundwater flow model using these two independent tests utilizing different measurement parameters. Additional evaluation will continue as the tracer test concludes in May 2007 to assess the site groundwater contaminant flow direction and flow rate of the effluent groundwater releases to the Hudson River.

.2 Temporary Instruction (TI) 2515/166 - Pressurized Water Reactor Containment Sump Blockage

a. Inspection Scope

The inspectors performed the inspection in accordance with Temporary Instruction (TI) 2515/166, "Pressurized Water Reactor Containment Sump Blockage." The TI was developed to support the NRC review of licensee activities in response to NRC Generic Letter (GL) 2004-02, “Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors (PWR).” Specifically, the inspectors verified the implementation of the modifications and procedure changes were consistent with the proposed actions committed to in the GL response. The inspectors reviewed a sample of the licensing and design documents to verify that they were either updated or in the process of being updated to reflect the modifications. A sample of operational, surveillance, and maintenance procedures, and calculations were reviewed to verify that they were in progress of being updated to reflect the effects of the modifications and the new requirements for the containment sumps and debris generation sources. The inspectors performed a walkdown of the strainer installation for both the internal recirculation and vapor containment sumps while modification work was in-progress to ensure the changes were consistent and performed in accordance with the approved design change package. This walkdown also included changes completed and in-progress with respect to the installation of flow channeling and flow barriers within the containment structure.

b. Evaluation of Inspection Requirements

The TI requested the inspectors to evaluate and answer the following questions.

1. Did the licensee implement the plant modifications and procedure changes committed to in their GL 2004-02 response?

   The inspectors verified that the modifications related to the installation of the sump screens, flow channeling and flow barriers described in Entergy's response to Generic Letter 2004-02 were in-progress and planned for completion during the current refueling outage. The inspectors noted that the installed sump strainers significantly increased the net free area of strainers from the previous design for both the internal recirculation and vapor containment sumps. The inspectors determined that the licensee had a calculation of record established with respect to the evaluation of debris generation and were in the process of evaluating if further changes were required to the analysis. Additionally, the licensee’s review for the potential of clogging downstream components due to debris bypass was in-progress. The inspectors noted that applicable procedures were in the process of being updated to reflect the effects of the modifications. The inspectors also determined that Entergy had developed adequate procedure controls for configuration control of insulation used inside containment.

2. Has the licensee updated its licensing basis to reflect the corrective actions taken in response to GL 2004-02?

   The inspectors verified that the changes to the facility, as described in the Updated Final Safety Analysis Report (UFSAR), which were identified in Entergy's GL 2004-02 response were reviewed and documented in accordance with their 10 CFR 50.59 screening process. The inspectors determined that the modifications had been appropriately described in the FSAR update package submitted for the sump modifications.

The TI will remain open to allow for the review of portions of the GL response that have not been completed. Specifically, Entergy had not completed the downstream effects analysis or chemical effects analyses. The results of these analyses have the potential to impact the licensing bases and programmatic procedures, and may require additional modifications. Therefore, this inspection will be considered incomplete until the results are reviewed. The NRC had set a December 2007 deadline for the completion of these evaluations.

c. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

Exit Meeting Summary

On April 4, 2007, the inspectors presented the inspection results to Mr. James Comiotes and other Entergy staff members, who acknowledged the inspection results presented. Entergy did not identify any material as proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

F. Dacimo, Site Vice President
K. Polson, General Manager, Plant Operations
J. Ventosa, Director, Engineering
J. Comiotes, Director, Nuclear Safety Assurance
A. Williams, IP3 Operations Manager
A. Vitale, Site Operations Manager
T. Barry, Security Manager
J. Donnelly, Manager, Maintenance
P. Conroy, Manager, Licensing
B. Sullivan, Emergency Planning Manager
T. Jones, Licensing Supervisor
L. Lee, Systems Engineering Supervisor
T. Orlando, Manager, Design Engineering
J. Seaboldt, Shift Manager, Operations
P. Cloughhessy, Maintenance Rule Program Coordinator
P. Parker, Superintendent, Maintenance

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000286/2007002-01	NCV	Failure to include the intake structure trash racks within the scope of the Maintenance Rule monitoring program (Section 1R12)
05000286/2007002-02	NCV	Inadequate Procedure for Recirculation Sump Interference Removal (Section 1R17)
05000286/2007002-03	FIN	Inadequate Corrective Actions for Failure to Appropriately Monitor Service Water Intake Bay Level (Section 1R17)
05000286/2007002-04	NCV	Inadequate Procedure for Conduct of RTD Cross Calibrations (Section 1R20)
05000286/2007002-05	NCV	Inadequate Procedure for Control of Temporary Modification (Section 1R23)

Closed

06000286/2006-002-00	LER	Manual Reactor Trip as a Result of Arcing Under the Main Generator Between Scaffolding and Phase A & B of the Isophase Bus Housing (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures
3-SOP-RW-002, Rev 22: "Intake Structure Operation"
3-SOP-RW-001, Rev 29: "Circulating Water System Operation"
OAP-008, Rev 2: "Severe Weather Preparations"
OAP-48, Seasonal Weather Preparation, Rev. 03

Condition Reports

IP3-2007-00453	IP3-2007-00459	IP3-2007-00447	IP3-2007-00451
IP3-2007-00471	IP3-2007-00485	IP3-2007-00473	IP3-2007-00472
IP3-2007-00456	IP3-2007-00470	IP3-2007-00579	IP3-2007-00505
IP3-2007-00690	IP3-2005-05389	IP3-2003-00753	IP3-2003-00469

Drawings
9321-F-22633, Sheet 2: "Intake Structure Circulating Water Piping"
9321-F-10133, Rev 10: "Intake Structure Cross Sections"
9321-F-20263, Rev 26: "Flow Diagram Circulating Water"
9321-F-20113, Rev 11: "Intake Structure General Arrangement"

Miscellaneous
Unit 3 Circulating Water System 2006-2007 Annual Health Report
Unit 3 Service Water System 3rd Quarter 2006 Health Report
IPEC Maintenance Rule Basis Document, Service Water System, Rev 0, 12/31/05
PMCR 03-IS-IMD-0130: Change to Intake Structure Planned Maintenance schedule
System Description 24.0, Rev 5: "Service Water System"
System Description 22.0, Rev 4: "Intake Structure"

Section 1R04: Equipment Alignment

Procedures
3-COL-EL-5, Rev 29: "Diesel Generators"-
3-PT-M079B, Rev 34: "32 EDG Functional Test"
3-PT-Q134A, Rev 3: "31 RHR Pump Functional Test (RHR Cooling not in Service)"
3-COL-RHR-1, Rev 25: "Residual Heat Removal System"
3-COL-CC-1, Rev 27: "Component Cooling System"
3-SOP-RHR-001, Rev 37: "Residual Heat Removal System Operation"
3-PT-M079C, Rev 33: "33 EDG Functional Test"

Drawings
9321-F-27503, Rev 45: “Flow Diagram, Safety Injection System, Sheet No 2"
9321-F-27513, Rev 29: “Flow Diagram, Auxiliary Coolant System in PAB and FSB, Sheet No 1"
9321-F-27353, Rev 39: “Flow Diagram Safety Injection System, Sheet No 1"
9321-F-27203, Rev 29: “Flow Diagram Auxiliary Coolant System”

Condition Reports

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IP3-2005-04828	IP3-2007-01054	IP3-2007-01327	IP3-2005-01260
IP3-2007-01632			

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IP3-06-20814	IP3-04-06426	IP3-05-00441	IP3-05-01930
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Section 1R05: Fire Protection

Procedures
ENN-DC-161, Rev 1: “Transient Combustible Program”
SMM-DC-901, Rev 2: “IPEC Fire Protection Program”

Miscellaneous
Pre-Fire Plan 354, Rev 0; “Diesel Generator Building, Elevation 15 Feet”
Pre-Fire Plan 305, Rev 0; “Safety Injection Pumps/ Main Corridor- PAB”
Pre-Fire Plan 306B, Rev 0; “41' Primary Auxiliary Building”
Pre-Fire Plan 365/366, Rev 0; “Auxiliary Feedwater Building”
Pre-Fire Plan 307A, Rev 0; “55' Primary Auxiliary Building”
Pre-Fire Plan 303, Rev 0; “Containment Building”

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Section 1R06: Flood Protection Measures

Procedures
3-AOP-FLOOD-1, Rev. 2: “Flooding”

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IP3-2006-00239	IP3-2006-01254

Other Documents
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Section 1R07: Heat Sink Performance

Procedures
EN-DC-147, Rev 2: "Indian Point Units 2 & 3 Eddy Current Program"
0-HTX-400-GEN, Rev 1: "Eddy Current Inspection of Heat Exchanger Tubes"

Other Documents
Indian Point Energy Center Generic Letter 89-13 Service Water Program, Revision 1
Indian Point Component Performance Monitoring Plan, Heat Exchangers

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Procedures
ENN-DC-190, Rev. 1, Steam Generator Eddy Current Data Analysis
3-PT-R204, Rev. 0, Visual Examination of Reactor Vessel Bottom Mounted Instrumentation Penetrations for Leakage
ENN-NDE-10.01, Rev 2, VT-1 Examination
ENN-NDE-10.03, Rev 1, VT-3 Examination
MRS-SSP-1450-INT, Rev 3, RPV Head Penetration Inspection Tool Operation
IPEC-UT-248, Field Change 2, Manual UT Examination of SG Primary Nozzle inside radius corner.
IPEC-UT-247, Rev 0, , Manual UT Examination of SG Feedwater Nozzle inside radius corner.
ENN-NDE-9.04, Rev 1, UT Examination of Ferritic Piping Welds (ASME Section XI)
ENN-NDE-9.11, Rev 0, Manual Ultrasonic Through Wall and Length Sizing of Ultrasonic Indications in Reactor Pressure Vessel Welds
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IPEC-UT-210, Rev 1, Manual UT Examination of Vessel Welds
EN-DC-319, Rev 0, Inspection and Evaluation of Boric Acid Leaks
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Section 1R11: Licensed Operator Requalification Program

Procedures
E-0, Rev 21: “Reactor Trip or Safety Injection”
E-1, Rev 18: “Loss of Reactor or Secondary Coolant”
ES-1.1, Rev 17: “SI Termination”
ES-1.2, Rev 12: “Post-LOCA Cooldown and Depressurization”
GFO-1, Rev 1: “Generic Foldout Page”
3-AOP-480V-1, Rev 2: “Loss of Normal Power to any Safeguards 480V Bus”
3-ONOP-FP-1, Rev 21: “Plant Fires”
0-AOP-SEC-1, Rev 2: “Response to Security Compromise”

Section 1R12: Maintenance Effectiveness

Procedures
ENN-DC-171, Rev 2: “Maintenance Rule Monitoring”
3-PT-M13B1, Rev 12: “Rx Protection Logic Channel Functional Test, Reactor Power Greater than 35% - P8"
AP-55, Rev 5: “Preventive Maintenance Program”
EN-DC-337, Rev 1: “Living Preventive Maintenance Program”
EN-DC-324, Rev 0: “Preventive Maintenance Process”
3-SOP-RW-002, Rev 22: “Intake Structure Operation”

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IP3-05-25547	IP3-05-23794	IP3-05-20289	IP3-06-16822
IP3-06-15079	IP3-05-16093	IP3-04-13192	IP3-03-04347
IP3-06-12102	IP3-03-01913	IP3-03-12010	IP3-05-24562
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9321-F-20333, Sheet 1: “Flow Diagram Service Water System”
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113E301, Sheet 2, Rev 13: “Reactor Protection System Schematic”
113E301, Sheet 3, Rev 10: “Reactor Protection System Schematic”
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113E301, Sheet 6, Rev 6: “Reactor Protection System Schematic”
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9321-LL-31343, Sheet 8, Rev 22: “Schematic Diagram Supervisory Annunciator”
9321-F-20363, Sheet 1: “Flow Diagram Instrument Air System”
9321-F-22633, Sheet 2: “Intake Structure Circulating Water Piping”
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9321-F-20113, Rev 11: “Intake Structure General Arrangement”

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Section 1R13: Maintenance Risk Assessment and Emergent Work Control

Procedures
IP-SMM-WM-101, Revision 1: “On-Line Risk Assessment”
IP-SMM-WM-100, Revision 5: “Work Control Process”
3PT-R151, Rev 3: “Test of Appendix ‘R’ Alternate Feed to 31 and 32 Charging Pump”
3-AOP-FH-1, “Fuel Damage or Loss of SFP/Refueling Cavity Level,” Revision 2
CR-IP3-07-00501
MRS-SSP-2017-IPP/INT, “Indian Point Nozzleless Fuel Assembly Handling Tool,” Revision 0
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IP-SMM-AD-102, Rev 4: “IPEC Implementing Procedure Preparation, Review and Approval”
EN-OP-104, Rev 2: “Operability Determinations”
OAP-026, Rev 0: “Determination of Operability”
EN-LI-102, Rev 8: “Corrective Action Process”
3-PC-OL07E3, Rev 0: “Steam Generator Level Analog Calibration Channel III”

3-PC-OL07E2, Rev 1: “Steam Generator Level Analog Calibration Channel II”
3-PC-OL07E1, Rev 1: “Steam Generator Level Analog Calibration Channel I”
3-PT-M108, Rev 3: “RHR/SI System Venting”
SOP 31.1.4, Rev 9: “Gas Turbine 1 Fuel Oil Receipt”
0-CY-1810, Rev 5: “Diesel Fuel Oil Monitoring”
0-CY-3320, Rev 1: “Relative Density / Specific Gravity of Fuel Oil”
3-PT-R176, Rev 5: “Stroke Testing of SI System Check Valves SI 895A-D and SI-897A-D”
SI-SOP-SI-001, Rev 38: “Safety Injection System Operation”

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IP3-CALC-05-00949, Rev 0: “Nitrogen Gas Accumulation from Check Valve SI838D Leakage”
IP3-CALC-ESS-00263, Rev 1: “Instrument Loop Accuracy/Setpoint Calculation, Steam Generator Narrow and Wide Range Level”

Miscellaneous
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Section 1R17: Permanent Plant Modifications

Modifications
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ER IP3-06-3-005, “IP3 ECCS Sump Strainer Upgrade,” Revision 0

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IP3-2007-01059	IP3-2007-00867	IP3-2007-00865	IP3-2007-01520
IP3-2007-01650	IP3-2007-01655	IP3-2007-01658	IP3-2007-01662
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Drawings
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9321-F-27513, Rev 29: “Flow Diagram Auxiliary Coolant System in PAB and FSB, Sh 1"
9321-F-27193, Rev 45: “Flow Diagram Waste Disposal System, Sh 1"
9321-F-27353, Rev 39: “Flow Diagram Safety Injection, Sh 1"

9321-F-20123, “Intake Structure - General Arrangement,” Revision 10

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EN-DC-105, “Configuration Management,” Revision 0
ENN-DC-103, “Design Process,” Revision 1
ENN-DC-115, “ER Response Development,” Revision 6
OAP-031, “Control of Operator Aids,” Revision 0
ENN-DC-112, Rev 7: “Engineering Request and Project Initiation Process”
ENN-DC-117, Rev 4: “Post Modification Testing and Special Testing Instructions”
3-OSP-WDS-001, Rev 2: “RCS and Refueling Cavity Cleanup”
OAP-7, “Containment Entry and Egress,” Revision 10
3-AOP-SW-1, “Service Water Malfunction,” Revision 2
EN-LI-102, “Corrective Action Process,” Revision 8

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Miscellaneous
ENN-DC-116, Rev 5, Att 9.1: “ER Response Installation, Pre-Installation Checklist”
IP-SMM-HU-102, Rev 0, Att 10.1: “Pre Job Briefing”
EN-RP-104, Rev 1, Att 9.12: “Personnel Contamination Event Record” (2 copies)

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OAP-024, Rev 2: “Operations Testing”
3-PT-M13B1, Rev 12: “Rx Protection Logic Channel Functional Test, Reactor Power Greater than 35% - P8"
3-PT-R160A, Rev.9: “31 EDG Capacity Test”
3-PT-CS029, Rev. 5; “Reactor Head Vent Valves Test”

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WO-IP3-05-15768
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3-PT-R013, “Recirculation Pumps in-Service Test,” Revision 19
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Ultrasonic test examination record 06UT053

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3-POP-4.2, Rev 23: "Operation Below 20% Przr Level with Fuel in the Reactor"
SOP-SG-005, Rev 1: "RCS Nozzle Dam Operation and Control"
3-SOP-RP-020, Rev 27: "Draining the Refueling Cavity"
3-SOP-RHR-001, Rev 37: "Residual Heat Removal System Operation"
0-NF-203, Rev 3: "Internal Transfer of Fuel Assemblies and Inserts"
3-REF-003-GEN, Rev 1: "Indian Point Unit 3 Refueling Procedure"
OAP-007, Rev 10: "Containment Entry and Egress"
3-POP-4.2, Rev 23: "Operation Below 20% Pressurizer Level with Fuel in the Reactor"
3-POP-4.1, Rev 25: "Operation at Cold Shutdown"

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Miscellaneous
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Section 1R22: Surveillance Testing

Procedures
3-PT-Q137, Rev 4: "Containment Building Inspection"
PFM-22A, Rev 7: "In-Service Test Program #9"
3PT-R160B, "32 EDG Capacity Test," Revision 9
3-SOP-EL-001, "Diesel Generator Operation," Revision 37

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Miscellaneous
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Section 1R23: Temporary Modifications

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EN-DC-136, Rev 0: “Temporary Modifications”

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IP3-2006-00879

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Section 1EP6: Drill Evaluation

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IP-EP-120, Rev 2: “Emergency Classification”
IP-EP-410, Rev 3: “Protective Action Recommendations”
IP-EP-AD1, Rev 1: “Maintaining Emergency Preparedness”

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Procedures
EN-LI-114, Revision 2: “Performance Indicator Process”
NEI 99-02, Rev. 4: “Regulatory Assessment Performance Indicator Guideline”
0-CY-2765, “Coolant Activity Limits - Dose Equivalent Iodine / Xenon and Average Energy (E-Bar),” Revision 1
EN-LI-114, Attachment 2, “NRC Performance Indicator Technique Sheet,” Revision 0, First Quarter 2006 to Fourth Quarter 2006

Section 4OA2: Identification and Resolution of Problems

Procedures
3-POP-4.2, Rev 23: “Operation Below 20% Przr Level with Fuel in the Reactor”
SOP-SG-005, Rev 1: “RCS Nozzle Dam Operation and Control”

Miscellaneous
3-SOP-RP-020, Rev 27: “Draining the Refueling Cavity”
0-SYS-014-GEN, Revs 4 and 5: “Scaffolding Construction and Control”

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Section 4OA5: Other Activities

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0-SYS-404-GEN, Installation of Insulating Materials for All Plant Piping and Equipment, Rev. 0
OAP-007, Containment Entry and Egress, Rev. 11
3-ONOP-WDS-1, Abnormal Containment Sump Levels, Rev. 8
3PT-R013, Recirculation Pumps Inservice Test, Rev. 19
3-Graph-TC-19, Recirculation Sump-Level/Volume, Rev. 1
3-Graph-TC-20, Containment Sump-Level/Volume, Rev. 0

Miscellaneous
IP3-CALC-RHR-00104, RHR Pump NPSH During Recirculation, Rev. 1
IP3-CALC-SI-02430, NPSHA/NPSHR for SI Internal Recirculation Pumps 31 & 32, Rev. 2
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Schematic of Injection Well Location and Design by GZA, January 31, 2007
Preliminary List of Sites Testing Positive for Fluorescein Dye in the Unit 2 Dye Introduction, Cathy Aley, Ozark Underground Laboratory, March 19, 2007

LIST OF ACRONYMS

ADAMS	agencywide documents and management system
ALARA	as low as reasonably achievable
ANS	alert notification system
AFW	auxiliary feed water
CAP	corrective action program
CCR	central control room
CEDE	committed effective dose equivalent
CFR	Code of Federal Regulations
CR	condition report
DEC	Department of Environmental Conservation
ECCS	emergency core cooling system
EDG	emergency diesel generator
EPRI	Electric Power Research Institute
ESSAP	Education, Environmental Site Survey and Assessment Program
°F	Fahrenheit

GL	generic letter
IMC	inspection manual chapter
IN	information notice
IP2	Indian Point Nuclear Generating Unit 2
IP3	Indian Point Nuclear Generating Unit 3
IPE	individual plant examination
LER	licensee event report
mrem	millirem
MSPI	mitigating system performance index
MW	monitoring well
NI	nuclear instrument
NCV	non-cited violation
NEI	Nuclear Energy Institute
NPSH	net positive suction head
NPSHA	net positive suction head available
NPSHR	net positive suction head required
NRC	Nuclear Regulatory Commission
ODCM	offsite dose calculation manual
ORISE	Oak Ridge Institute for Science and Education
PARS	publicly available records
PI	performance indicator
psig	pounds per square inch gage
PWT	post-work test
QPTR	quadrant power tilt ratio
RETS	radiological effluents technical specifications
RHR	residual heat removal
RP	radiation protection
RW	recovery well
RWP	radiation work permit
SDP	significance determination process
SE	safety evaluation
SFP	spent fuel pool
SI	safety injection
SSC	systems, structures, components
T	temperature
TI	temporary instruction
TS	technical specifications
UE	unusual event
URI	unresolved item
UFSAR	updated final safety analysis report
WO	work order